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Columbia Generating Station
Energy Northwest/Framatome ANP
Transition Cycle Licensing

Presentation to NRC Staff

by

Mike Humphreys

Manager, Engineering

Energy Northwest

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Meeting Agenda

- Purpose
- Background
- Transition Approach, Analyses and Methodology
- Analysis of Co-Resident SVEA 96 Fuel
- Fuel Corrosion Assessment
- Licensing Submittals and Schedule
- Conclusions

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Purpose

- Communicate to the Staff our transition plans and status including the licensing submittals required to effect a safe and efficient Cycle 17 transition
- Describe the transition process from Westinghouse SVEA-96 fuel to Framatome ANP ATRIUM™-10 Fuel
- Update the Staff on fuel corrosion status
- Discuss the licensing activities and schedule required to support Cycle 17 operation beginning in June 2003

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Background

- Columbia Generating Station is currently operating in Cycle 16 with a full core (764 assemblies) of SVEA-96 fuel. The next refueling outage will begin in mid-May 2003
- The Cycle 17 core will consist of 280 fresh ATRIUM™-10, 12 fresh SVEA-96 and 472 burned SVEA-96 fuel assemblies. Restart will begin in mid-June 2003
- Both assembly designs are a 10x10 matrix
  - ATRIUM™-10: 9 'center' rods replaced by a channel box leaving 91 fuel rods of which 8 are part length rods; 8 spacers per assembly, a debris resistant lower tie plate, average enrichment is 3.92% U_{235}
  - SVEA-96: 4 center rods replaced by a water cross leaving 96 fuel rods in four distinct 5X5 subassemblies; 6 spacers per subassembly, average enrichment is 3.86% U_{235}

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ATRIUM™-10 Assembly Layout

- Refer to a full page slide at the end of the handout package for the assembly layout drawing

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SVEA-96 Fuel Assembly

- Refer to a full page slide at the end of the handout package for the fuel assembly drawing

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Transition to ATRIUM™-10 Fuel

- Transition Approach
- Transition Cycle Analyses
- Analysis Methodology

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Transition Cores
General Approach

- Minimize changes to current plant licensing basis
- Evaluate the introduction of ATRIUM™-10 fuel per the requirements of 10 CFR 50.59
  - Similar to approach used for any plant change
  - Similar to approach used for each reload core design (except for scope)
- Identify plant safety analyses potentially affected by a fuel or core design change
- Assess impact of fuel design change on plant safety analyses and repeat analyses if the conclusions of the analysis are potentially affected

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Transition Cores
General Approach *(Continued)*

- Technical Specification changes generally limited to
  - References to NRC-approved methods used to determine thermal limits specified in the COLR
  - MCPR safety limit based on Framatome methods
  - Design features
- COLR thermal limits are determined for the transition core based on analyses using NRC-approved methods

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Transition Core Approach
Establish Current Licensing Basis

- Licensing basis consists of all analyses performed to demonstrate that regulatory requirements are met
- Columbia's licensing basis is defined in documents such as
  - FSAR
  - Technical Specifications
  - Core Operating Limits Report (COLR)
  - Licensee Controlled Specifications (Technical Requirements Manual)
  - Cycle Reload Licensing Reports which include
    - Extended Operating Domain analyses (e.g. increased core flow operation)
    - Equipment Out-of-Service analyses (e.g. turbine by-pass valve OOS)
  - LOCA Analysis Reports

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Transition Core Approach
Disposition of Events

- Review all event analyses identified in the FSAR and in other applicable licensing analysis reports
- Analyses are dispositioned as
  - Not impacted by the change in fuel or core design
  - Bounded by the consequences of another event
  - Potentially limiting - analyze using Framatome methodology
- Rated and off-rated conditions are considered
- Potentially limiting events are evaluated generically or reanalyzed each cycle using NRC-approved methodology

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Transition Cycle Analyses

- Results from the disposition of events define the required transition cycle safety analyses for each of the following areas
  - Thermal-Hydraulic Analyses
  - Anticipated Operational Occurrence (AOO) Transient Analyses
  - Accident Analyses
  - Special Analyses
- A preliminary disposition of events has been completed for Columbia

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Transition Cycle Analyses
Thermal-Hydraulic Analyses

- Hydraulic compatibility
  - Core pressure drop
  - Core bypass flow
  - Thermal (MCPR) margin
  - Water channel voiding
- Minimum Critical Power Ratio (MCPR) safety limit
  - Protects fuel from boiling transition during both normal operation and during AOOs
  - Analysis performed on a cycle-specific basis
  - Transition cycle analysis explicitly models mixed core configuration
- Slow flow excursion analysis
  - Establish or verify $MCPR_f$ limits
  - Analysis addresses both Framatome and coresident fuel

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Transition Cycle Analyses
AOO Analyses

- Analyses are performed to establish power dependent thermal limits (MCPR, LHGR) that protect against fuel failure during an AOO
- Expected limiting AOOs and analytical methodology applied for analysis

Control rod withdrawal error	Neutronic Methods
Loss of feedwater heating	Neutronic Methods
Load rejection without bypass	Transient Methods
Turbine trip without bypass	Transient Methods
Feedwater controller failure	Transient Methods
Recirculation Flow Run-up	TH Methods

Transition Cycle Analyses
Accident Analyses

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- Analyses are performed to confirm that radiological release are within 10 CFR limits during postulated accidents
- Expected limiting accidents and analytical methodology applied for analysis
  - Loss of coolant — LOCA Methods
  - Control rod drop — Neutronic Methods
  - Fuel assembly misload — Neutronic Methods
  - Fuel handling accident — Neutronic Methods
- LOCA analysis establishes a MAPLHGR limit for each fuel type to ensure that 10 CFR 50.46 criteria are satisfied
  - LOCA break spectrum analysis performed for introduction of ATRIUM™-10
  - Limiting LOCA heatup analysis performed for each cycle

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Transition Cycle Analyses
Special Analyses

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- Analyses are performed to demonstrate the capability of the plant to meet specific regulatory requirements or industry codes
- Expected limiting analyses and analysis methodology
  - Shutdown margin — Neutronic Methods
  - Standby liquid control — Neutronic Methods
  - Stability — Neutronic Methods
  - ASME overpressure protection — Transient Methods
  - Appendix R Fire Protection — LOCA Methods

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BWR Safety Analysis Methodology

- Potentially limiting events are analyzed using Framatome ANP BWR safety analysis methodology approved by the NRC
- No new methodology submittals are expected to be required for the transition to ATRIUM™-10 reload fuel at Columbia

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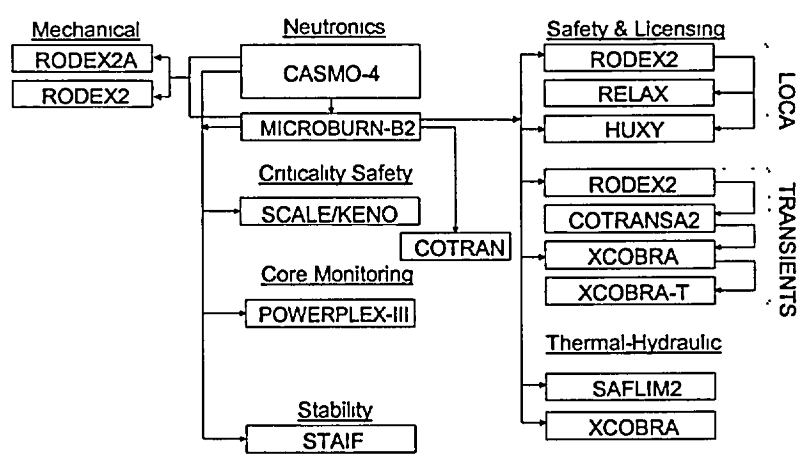
BWR Safety Analysis Methodology



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Thermal-Hydraulic Analysis Methodology
Major Computer Codes

- XCOBRA [XN-NF-79-59(P)(A)]
  - Predicts the steady-state performance of BWR cores at various operating conditions and power distributions
- SAFLIM2 [ANF-524(P)(A) Revision 1 and Supplements 1 and 2]
  - Used to determine an acceptable MCPR safety limit for a core design

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Neutronic Analysis Methodology
Major Computer Codes

- CASMO-4/MICROBURN-B2 [EMF-2158(P)(A)]
  - Assess impact on thermal limits during localized or quasi-steady-state events
- COTRAN [XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2]
  - Determine core power response during a control rod drop accident
- STAIF [EMF-CC-074(P)(A) Volume 4]
  - Calculate core and channel decay ratio (frequency domain)

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Transient Analysis Methodology
Major Computer Codes

- RODEX2 [XN-NF-81-58(P)(A) Revision 1 and Supplements 1 and 2]
  - Calculate gap conductance for core and hot channel
- XCOBRA [XN-NF-79-59(P)(A)]
  - Determine hot channel initial active flow
- COTRANSA2 [ANF-913(P)(A) Revision 1 and Supplements 2, 3, 4]
  - Calculate reactor system and core response during transient events
- XCOBRA-T [XN-NF-84-105(P)(A) Volume 1 and Supplements 1, 2]
  - Calculate ΔCPR for each hot channel in the core

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LOCA Analysis Methodology
Major Computer Codes

- RODEX2 [XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and2]
  - Determine initial fuel characteristics
- RELAX [EMF-2361(P)(A)]
  - Determine reactor system response during the blowdown phase
  - Determine hot channel response during the blowdown phase
  - Determine reactor system response during the refill and reflood phases (time of core reflood)
- HUXY [XN-CC-33(P)(A) Revision 1]
  - Calculate PCT and metal-water reaction rate at limiting axial plane

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LOCA Analysis Methodology
Changes

- Methodology Changes for Columbia
  - LOCA analysis for Columbia is the first application of the EXEM BWR-2000 methodology
  - Minor changes to the methodology were identified during the implementation of the methodology for Columbia
  - The effects of these changes will be reported per the requirements of 10 CFR 50.46

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Transition Core Approach
Summary

- The transition process is well understood by Energy Northwest and Framatome ANP
- Energy Northwest successfully transitioned from the initial GE 8X8 core to Exxon 8X8 fuel in 1986 and from ANF 9X9 to the SVEA-96 design in 1996
- Energy Northwest and Framatome ANP have proven procedures, methods, and technology to support a smooth transition to ATRIUM™-10 fuel

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Analyses of Co-Resident SVEA-96 Fuel

- Mixed Core Analysis Approach-Licensing Approach
- SVEA-96 Fuel Considerations
  - Geometry
  - Neutronic
  - Thermal-Mechanical
  - Thermal-Hydraulic
  - CPR Correlation

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Mixed Core Analyses
Licensing Approach

- Analyses are performed to confirm that all design and licensing criteria are satisfied
- Analyses explicitly include each fuel type in the core
  - Performed using generically approved methodology
  - Cycle-specific core loading considered
  - Input data appropriate for each fuel type used
  - Models were developed for SVEA-96 that are consistent with and in accordance with Framatome ANP methods
  - Required geometric, neutronic and thermal hydraulic data for the SVEA-96 fuel was formally transmitted to Framatome by Energy Northwest
- Thermal limits are established for each fuel design
- During operation each fuel type is monitored against applicable limits

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Analysis for SVEA-96 Fuel
Geometric Data

- Detailed description of the geometry of the SVEA-96 fuel is used in the design and licensing analyses
- SVEA-96 fuel design drawings and reports provided the data for the geometric models used in all the mixed core calculations including
  - Mechanical compatibility analyses
  - Thermal-mechanical analyses
  - Thermal-hydraulic analyses
  - Neutronic analyses
- Data provided by Energy Northwest is at the same level of detail as used in analyses for Framatome ANP fuel

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Analysis for SVEA-96 Fuel
Neutronic Characteristics

- Fuel nuclear characteristics (enrichment, Gd concentrations) are required for
  - Neutronic design of transition cores
  - Determine core reactivity characteristics for use in safety analyses
- Data provided by Energy Northwest is at the same level of detail as used in analyses for Framatome ANP fuel

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Analysis for SVEA-96 Fuel
Thermal-Mechanical Considerations

- Fuel rod thermal-mechanical characteristics are required for the RODEX2 analyses to determine pellet-to-cladding gap coefficients used in the transient analyses
- The standard RODEX2 approach is used for the SVEA-96 fuel with the exception that an annealed cladding model is used
- Thermal-mechanical limits (e.g. LHGR limits) for SVEA-96 fuel will continue to be based on Westinghouse analyses; no Framatome analysis will be performed
- Data provided by Energy Northwest is at the same level of detail as used in analyses for Framatome ANP fuel

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Analysis for SVEA-96 Fuel
Thermal-Hydraulic Characteristics

- Fuel hydraulic characteristics (e.g., pressure loss coefficients) are required for
  - Core design analyses (core flow distribution, MCPR)
  - Thermal Hydraulic compatibility (pressure drop, core bypass flow)
  - Plant transient analyses (MCPR)
- Thermal-hydraulic analyses explicitly address mixed core issues
- Hydraulic models for SVEA-96 fuel are developed from
  - Geometric description of the fuel assembly
  - Loss coefficients, pressure drop and flow data provided by Energy Northwest
- Relative pressure drop comparisons between ATRIUM™-10 and SVEA-96 fuel are consistent with pressure drop data measured by Framatome.

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Critical Power Correlation
SVEA-96 Fuel

- An approved Framatome CPR correlation is applied to the SVEA-96 as described in EMF-2245(P)(A) [*Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel*]
  - Indirect process-used when experimental data IS NOT available
  - Direct process- used when experimental data IS available
- The direct process from EMF-2245(P)(A) was used by Energy Northwest to determine CPR correlation constants for the SVEA-96 fuel.
  - Preferred approach
  - uses available experimental critical power test data to directly determine the additive constants for the SVEA-96 fuel
  - determines the behavior of the correlation over the range of conditions used to obtain the data

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Critical Power Correlation
SVEA-96 Fuel (*continued*)

- The direct process, performed by Energy Northwest, includes
  - Determining the additive constants for the SVEA-96 fuel
  - Determining the standard deviation for the experimental critical power ratio (ECPR) and the additive constants using an approved CPR correlation
  - Evaluating the application to determine that no unexpected trends are observed
  - Identifying the correlation parameter range applicable for SVEA-96 fuel
- The guidelines of GL 83-11 Supplement 1 have been met
  - Procedures
  - Training and qualification
  - Comparison calculations
  - Q/A-Change control process

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Critical Power Correlation

SVEA-96 Fuel - Statistical Results (Preliminary)

- Additive constants for SVEA-96 fuel have been developed for Framatome's SPCB-A10 CPR correlation
- The results of the analysis are

Mean ECPR	0.992
ECPR Standard Deviation	0.045
Additive Constant Standard Deviation	0.040

- Parameter range of correlation covers the expected range of operation (pressure, mass flow and enthalpy)

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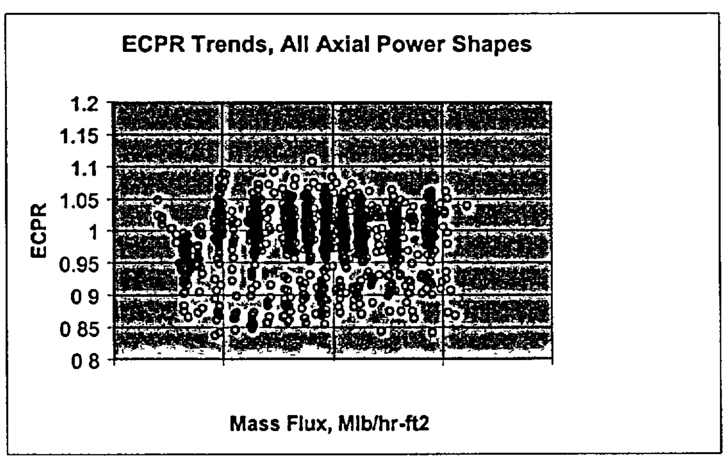
Critical Power Correlation

SVEA-96 Fuel - Trending Evaluation (Preliminary)



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SVEA-96 Critical Power Correlation
Subassembly Flow Distribution

- It is assumed there is no flow between the SVEA-96 subassemblies
- A non-uniform subassembly power distribution can result in non-uniform or mismatched subassembly flows
- CPR calculations for SVEA-96 fuel will be performed on a subassembly (5x5 array) basis
- Since the SPCB correlation uses an assembly average flow, an adjustment to the flow used in the calculation is needed when non-uniform subassembly flows exist

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SVEA-96 Critical Power Correlation
Subassembly Flow Distribution (*continued*)

- Given the total assembly flow and subassembly power the flow through each subassembly will be determined using approved thermal-hydraulic methodology
- The subassembly power is determined using the local pin power distribution
- CPR is calculated on a subassembly basis using the subassembly flow and power distribution

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SVEA-96 Critical Power Correlation
Application for Fresh SVEA-96 Fuel

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- For Columbia Generating Station Cycle 17, 12 fresh SVEA-96 assemblies will be loaded
- EMF-2245(P)(A) [*Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel*] is intended for application only to exposed co-resident fuel; lead fuel assemblies are not subject to this restriction
- Lead test assembly MCPR restrictions will therefore be placed on these 12 fresh assemblies
  - Loaded and operated so they will be non-limiting MCPR assemblies

Analysis of SVEA-96 Fuel Summary

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- Cycle design and licensing evaluations explicitly consider the SVEA-96 fuel
- Limits are established for each fuel type
- Fuel characteristics and models are developed and applied consistent with approved methodology

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Fuel Corrosion Assessment

- Status/Summary
  - Three previous updates to NRC personnel on corrosion
    - Teleconference in February (NRC)
    - Presentation to Region IV (May 24, 2002)
    - Site briefing to Branch Chief E (August 20, 2002)
  - Today's meeting: discuss licensing approach
- Corrosion Root Cause
- Corrective Actions
- Licensing Requirements
- Licensing Approach for the Transition Cycles
- Corrosion Ongoing Investigation & Monitoring
- Conclusions

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Fuel Corrosion Summary Status

- R15 fuel inspections revealed corrosion greater than what has been previously observed at Columbia Generating Station
  - Nodule formation
  - Oxide measurements higher than previous CGS experience
  - More CRUD than previous CGS experience
- Energy Northwest has proactively pursued a full understanding of the corrosion problem
  - Created a Task Force, assigned dedicated resources including external, expert consultants
  - Includes extensive review of industry experience, US & world wide
  - Investigation is being finalized, carried on by a Project team
- We have reached a good understanding of the root cause
  - Root cause investigation has identified a unique event in Cycle 15 that initiated nodule formation and accelerated corrosion
  - Corrective actions are in place to preclude a recurrence of the event
- There are no indications of fuel failures in Cycle 16
- Cycle 17 licensing issues are being addressed

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Draft Root Cause Results

- Primary Root Cause
  - During a specific period late in Cycle 15, one or more substances either deposited on the fuel surfaces or caused chemical reactions near the fuel surfaces, leading to nodule formation and accelerated fuel corrosion
    - This root cause remains preliminary, in that the substance or substances in question have not been positively identified and not all of the mechanisms involved are fully understood
    - However, concurrent trends in several water chemistry parameters during this period constitutes evidence that provides confidence in the primary root cause stated above
- Major Contributing Causes
  - Condenser Leak
  - Degraded Condensate Filter Demineralizer (CFD) Performance
- Minor Contributing Causes
  - Fe & Zn Injection
  - Presence of Copper (provides risk of CILC failure)

Corrective Actions

- Action 1: Implement Optimized Fe & Zn Targets
  - Balance fuel corrosion concerns with concerns about radiological hazard
- Action 2: Perform Copper Reduction Conceptual Study
  - Postponed-awaits results of ongoing investigations
- Action 3: Eliminate Condenser Leak
  - Condenser Reliability Project established January 2002
    - Independent review of condenser reliability is complete
    - Tube re-plugging planned for R16
    - Fill test/leak test in R16 is funded and feasibility is being evaluated
- Action 4: Optimize CFD Performance
  - Independent review of CFD and RWCU systems is complete
  - Pre-coat optimization recommendations from the review are being implemented
  - CFD effluent monitoring has been increased
  - Attention to CFD septa change-outs has been increased
  - Significant improvement in CFD performance has been demonstrated

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Licensing Requirements

- 10 CFR Part 50, Appendix A, *General Design Criteria for Nuclear Power Plants*, Criterion 10, *Reactor Design*
  - The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences
- Safety reviews intended to assure compliance with GDC-10 are performed in accordance with the USNRC Standard Review Plan (SRP)

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Licensing Requirements (Cont)

- SRP 4.2, *Fuel System Design*, Section I, Areas of Review
  - The objectives of the fuel system safety review are to provide assurance that
    - (a) the fuel system is not damaged (fuel rods do not fail) as a result of normal operation and anticipated operational occurrences
    - (b) fuel system damage is never so severe as to prevent control rod insertion when it is required
    - (c) the number of fuel rod failures is not underestimated for postulated accidents, and
    - (d) coolability is always maintained
- These requirements are implemented in two ways
  - Application of NRC-approved fuel vendor methodologies in fuel design
  - Programmatic controls that preclude fuel failure. Example: CILC failure risk
    - Careful control of reactor water chemistry
    - Use of cladding material that is resistant to nodule formation and corrosion

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Cycle 17 Licensing Approach

ATRIUM™-10 Fuel

- Apply Framatome's NRC-approved methodology for fuel design and analysis
  - Corrective actions are in place to preclude a recurrence of the Cycle 15 event and therefore no accelerated corrosion is expected in Cycle 17
- CILC failure risk is minimized
  - 10x10 design inherently has lower heat flux at the fuel rod surface (heat flux is an important parameter in the CILC failure process)
  - Cladding choices for ATRIUM-10 fuel are optimized to balance CILC failure risk with high burnup performance
    - LTP cladding on UO_2 rods
    - Late ß-quenched cladding on Gd rods
  - Corrective actions are in place to preclude a recurrence of the Cycle 15 event. Therefore, no nodule formation or accelerated corrosion is expected in Cycle 17

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Cycle 17 Licensing Approach (Cont)

SVEA-96 Fuel

- Apply Westinghouse's NRC-approved fuel rod mechanical design methodology for fuel design and analysis
  - SVEA-96 fuel loaded Cycle 16 and Cycle 17
    - These fuel batches were not subjected to the water chemistry environment that initiated accelerated corrosion in Cycle 15
    - Corrective actions will preclude a recurrence of the Cycle 15 event and therefore no accelerated corrosion is expected in Cycle 16 or Cycle 17
  - SVEA-96 fuel in Cycle 17 that was in the Cycle 15 core
    - The Cycle 16 Startup Evaluation completed by Westinghouse in June 2001 concluded that the fuel will operate within the fuel rod design bases identified in the topical report for Westinghouse's NRC-approved fuel rod mechanical design methodology
    - A Cycle 16 Operability Determination completed in April 2002 confirmed that the conclusions of the Startup Evaluation remain valid
    - The conclusions of both the Startup Evaluation and the Operability Determination are applicable to Cycle 17 (the analysis was not based on fuel rod burnup)
    - Westinghouse has verbally indicated concurrence with this position

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Cycle 17 Licensing Approach (Cont)

SVEA-96 Fuel (continued)

- CILC failure risk is minimized
  - SVEA-96 fuel loaded Cycle 16 and Cycle 17
    - These fuel batches were never subjected to the environment that initiated nodule formation and accelerated corrosion in Cycle 15
    - Corrective actions will preclude a recurrence of the Cycle 15 event and therefore no accelerated corrosion is expected in Cycle 16 or Cycle 17
    - 10x10 design inherently has lower heat flux at the fuel rod surface (heat flux is an important parameter in the CILC failure process)
    - SVEA-96 cladding is CILC-resistant late ß-quenched
  - SVEA-96 fuel in Cycle 17 that was in the Cycle 15 core
    - This batch of fuel is now at a burnup beyond the exposure at which the risk of CILC failure is highest (bundle power/heat flux is too low)
    - 10x10 design inherently has lower heat flux at the fuel rod surface
    - SVEA-96 cladding is CILC-resistant late ß-quenched
    - Corrective actions will preclude a recurrence of the Cycle 15 event and therefore the accelerated corrosion is not expected to continue

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Ongoing Investigation/Monitoring

- Inspect fuel during upcoming outages
  - R16 inspections (Westinghouse SVEA-96 fuel)
  - R17 (ATRIUM-10 fuel)
  - R18 (ATRIUM-10 fuel)
  - Ongoing outage inspections are expected beyond R18
- Finalize investigation and continue monitoring
  - Finalize fuel corrosion root cause report
  - Continue tracking applicable industry events
  - Monitor important water chemistry parameters
  - Track corrective actions in place to assure no repeat of the Cycle 15 event

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Fuel Corrosion-Conclusions

- Fuel Corrosion corrective actions are in place to preclude a recurrence of the Cycle 15 event
  - No accelerated corrosion expected in Cycle 17
- Programmatic controls assure that risk of CILC failure is low
  - 10x10 fuel provides lower average heat flux
  - Cladding materials mitigate nodule formation
  - Cycle 15 fuel with nodules is beyond the point of peak CILC risk
- Transition cycle licensing
  - Employ Framatome NRC-approved methodology for ATRIUM-10 fuel
  - SVEA-96 fuel will operate within the fuel rod design bases identified in the topical report for Westinghouse's NRC-approved fuel rod mechanical design methodology
- Finalize fuel corrosion investigation activities
  - Finalize the root cause report
  - Ongoing monitoring of critical water chemistry parameters
  - Ongoing fuel inspections in R16 and future outages

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Licensing Submittals and Schedule

- Licensing Submittals
  - Amendment Request dated September 3, 2002 proposed Technical Specification changes to
    - Add depleted U as an acceptable fuel material
    - Update the list of approved methodologies
    - Implement TSTF-463 to simplify the document references in the Technical Specification
  - Future Amendment Request to propose a revised Minimum Critical Power Safety Limit is planned for December 2002
  - A 10 CFR 50.46 Report describing methodology changes and effects is planned within 30 days of plant restart
- Requested Approval Schedule
  - Approval and issue of Licensing submittals is requested on or before May 31, 2003 in order to support restart for Cycle 17 operation in mid June.

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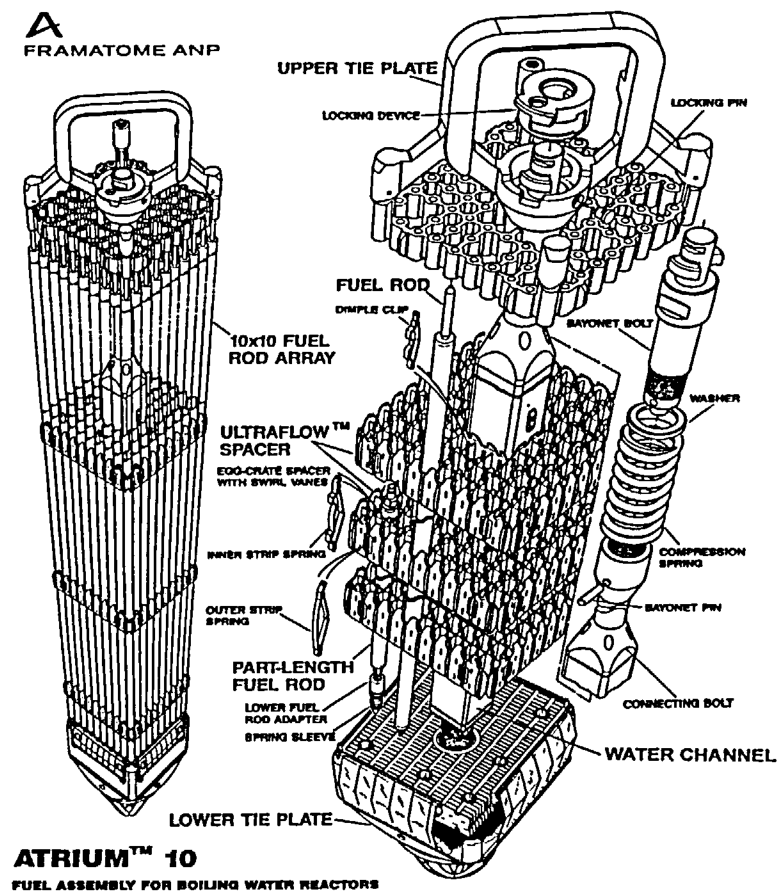
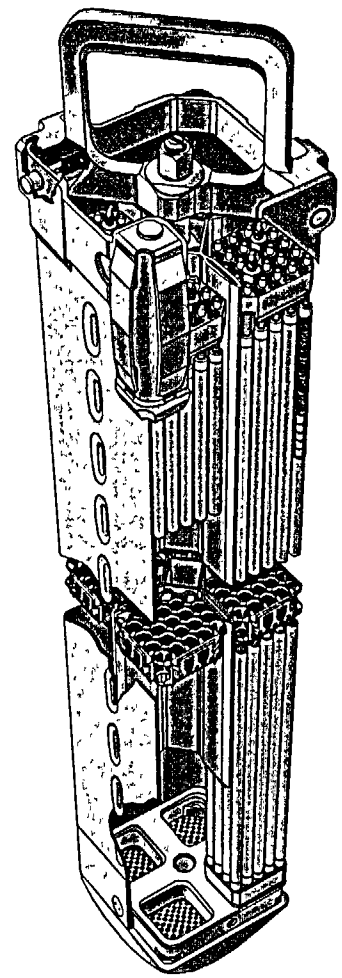
CONCLUSIONS

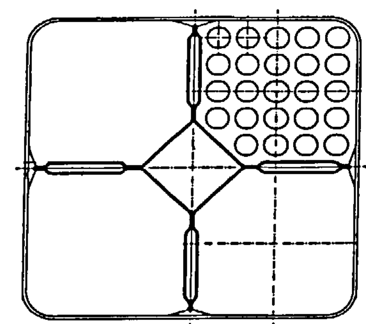
- The process required to transition from SVEA-96 to ATRIUM™-10 as the Cycle 17 reload fuel is well understood
- No changes in approved Framatome ANP methodology are required to address the SVEA-96
- Fuel corrosion issues are well understood and actions have been taken to ensure continued reliable fuel operation
- The required licensing actions are identified and no unusual issues are apparent

ATRIUM™-10 Assembly Layout



SVEA-96 Fuel Assembly





Assembly
Cross Section

SVEA-96 Fuel Assembly